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Nordic Forum for Generation IV Reactors, Status and activities in 2012

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December 2012

Abstract

The Nordic-Gen4 (continuation from NOMAGE4) seminar was this year hosted by DTU Nutech at Risø, Denmark. The seminar was well attended (49 participants from 12 countries). The presentations covered many aspects in Gen-IV reactor research and gave a good overview of the activities within this field at the various institutes and universities. Participants had the possibility to visit the Danish Decommissioning site.

Key words

Nordic-Gen4, seminar, Risø

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Nordic-Gen4 activities 2012

Nordic Forum for Generation IV Reactors

Status and activities in 2012

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1. INTRODUCTION

Generation IV reactors (Gen IV) are a set of theoretical nuclear reactor designs currently being researched. Most of these designs are generally not expected to be available for commercial construction before 2030. Demonstration of concepts are however foreseen around 2015-2020. Current reactors in operation around the world are generally considered second- or third-generation systems, with the first-generation systems having been retired some time ago. Relative to current nuclear power plant technology, the claimed benefits for 4th generation reactors include 1) sustainability, 2) increased safety, 3) better use of the nuclear fuel, 4) high efficiency and better economics, 5) minimal waste production, 6) the ability to consume existing nuclear waste in the production of electricity, 7) increased proliferation resistance.

Many reactor types were considered initially; however, the list was downsized to focus on the most promising technologies and those that could most likely meet the goals of the Gen IV initiative. At present one focuses on six different types of reactors: 1) very-high-temperature reactor (VHTR), 2) supercritical-water-cooled reactor (SCWR), 3) molten salt reactor (MSR), 4) gas-cooled fast reactor (GFR), 5) sodium cooled fast reactor (SFR), 6) lead-cooled fast reactor (LFR). The VHTR has an open fuel cycle, the SCWR can have an open or closed fuel cycle, while the other concepts have closed fuel cycles. Because of their high operational temperature, several of the above concepts (GFR, MSR, VHTR) can be used for the cogeneration of hydrogen. Internationally, research is organized through the Generation IV International Forum (GIF), under the leadership of the US. The participation of Europe in GIF is through EURATOM, concentrating on the following concepts; GFR, SCWR, SFR and VHTR. Research into Generation IV reactors is also part of the activities of the European Sustainable Nuclear Energy Technology Platform (SNETP) which aims at reducing the green-house gas emissions by 20 % by 2020 (compared to 1990), make 20% energy savings and include 20 % share of renewable energies in the total energy mix.

VHTR In the past (70’s and 80’s), the High Temperature Reactor (HTR) concept has already proved its viability in US, GB and Germany operating up to 950 °C. The VHTR represents a modern and highly evolved version of the original HTR design. To improve in a consistent manner the economic performances, e.g. thermodynamic cycle and hydrogen processes efficiency, this new concept must produce an outlet gas temperature above 1000 °C. In order to fulfil these two major requirements (higher operating temperature and hydrogen production in terms of technological challenges and safety), an important international R&D program is underway to establish the viability of the VHTR by 2010 and to optimize its design features as well as operating parameters by 2015.

SCWR: The SCWR enables a thermal efficiency about one-third higher than current light-water reactors, as well as a simplification in the plant because the supercritical water can directly be used to drive the turbines (the steam conversion process is eliminated). The SCWR is a concept that uses supercritical water as the working fluid. SCWRs are basically light water reactors (LWR) operating at higher pressure and temperatures (500-600 °C). The SCWR is built upon two proven technologies, Light Water Reactors (LWRs), which are the most commonly deployed power generating reactors in the world, and supercritical fossil fuel fired boilers, a large number of which are also in use around the world.

MSR: Molten salt reactor systems (MSR) use liquid salts as a coolant and a fuel together. The system has a coolant outlet temperature in the range 700-800 °C, affording improved thermal efficiency. The main benefits of the MSR system are that it offers an integrated fuel cycle, embodying a burner/breeder reactor concept whilst taking advantage of the excellent heat transport properties of molten salt.

GFR: The high outlet temperature of the helium coolant used in the GFR system makes it possible to deliver electricity, hydrogen, or process heat with high efficiency. The outlet temperature would be of the order of 850 °C. The GFR uses a direct-cycle helium turbine for electricity generation, or can optionally use its process heat for thermochemical production of hydrogen. Through the combination of a fast spectrum and full recycle of actinides, the GFR minimizes the production of long-lived radioactive waste.

SFR: The SFR reactor concept is cooled by liquid sodium (at 550 °C) and fuelled by a metallic alloy of uranium and plutonium. The SFR's fast spectrum makes it possible to use available fissile and fertile materials (including depleted uranium) considerably more efficiently than thermal spectrum reactors with once-through fuel cycles. One of the design challenges of an SFR is the risk of handling Sodium, which reacts explosively if it comes into contact with water.

LFR: The lead-cooled fast reactor features a fast-neutron-spectrum lead or lead/bismuth eutectic (LBE) liquid-metal-cooled reactor with a closed fuel cycle. The LFR is cooled by natural convection with a reactor outlet coolant temperature of 550 °C, possibly ranging up to 800 °C with advanced materials. Long-term R&D is required to develop high-temperature corrosion resistant materials.

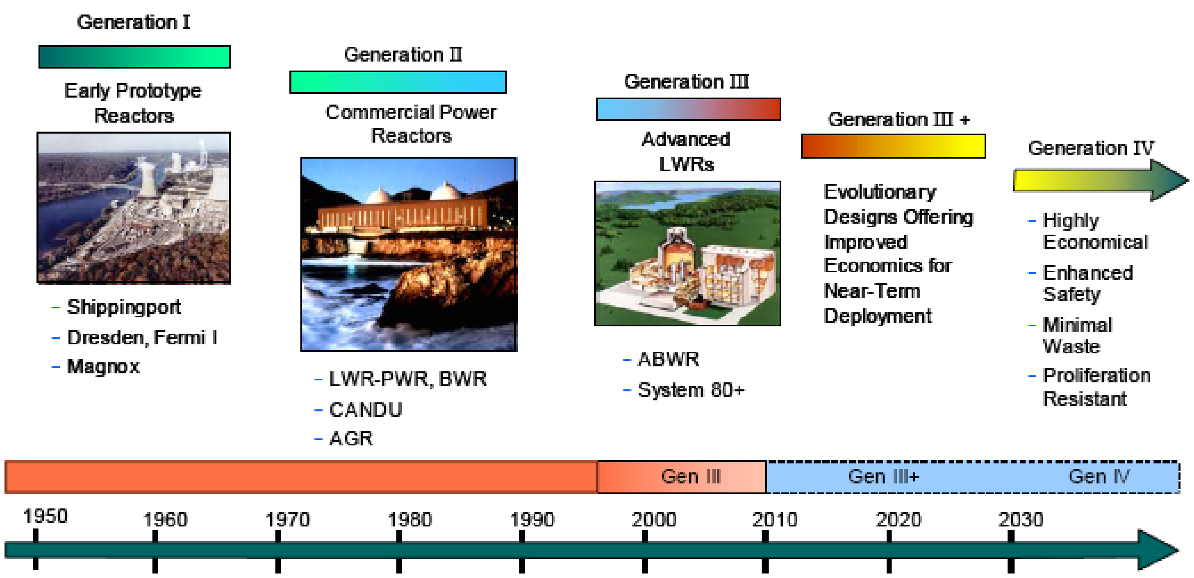

Figure 1: Time evolution of the different generations of reactor concepts.

Potential cladding and structural materials for the different GenIV systems are presented in Table 1.

Table 1 : Potential cladding and structural materials for the different GenIV systems [1-8]

System	Materials		
	Cladding	Core regions	Out of core regions
Gas-Cooled Fast Reactor System (GFR)	Ceramics Matrices of SiC, ZrC & TiN, ODS (Oxide Dispersion-Strengthened)	Ceramics Carbides SiC, ZrC Nitrides ZrN, TiN Oxides MgO $ZrYO_2$ Zr_3Si_2 ODS	Coated or non coated ferritic-martensitic or austenitic steels Nickel based super alloys ODS
Lead-Cooled Fast Reactor System (LFR)	Austenitic, ferritic-martensitic steel Coated cladding e.g. FeAl		
Molten Salt Reactor System (MSR)	-	Graphite Nickel-based alloys Ceramics	
Sodium-Cooled Fast Reactor System (SFR)	ODS (Metallic fuel) Ferritic-Martensitic steels (MOX-fuel)	Advanced austenitic steels	
Supercritical –Water-Cooled System (SCWR)	Austenitic, Ferritic-Martensitic steels ODS		Ferritic- Martensitic steels
Very-High-Temperature Reactor System (VHTR)	ZrC	Graphite Ceramics	Ni-Cr-W super alloys High temperature metal alloys

There is a strong link between the materials integrity and nuclear safety for all nuclear reactors. The operation conditions in new generation reactors will be more demanding and thus knowledge about materials behaviour and integrity under operation are critical. To evaluate and choose proper materials to be used for GenIV reactors, it is important to know the boundary limits for use of the materials under the specific operating conditions. Safety analysis is based on technological assessment where the materials integrity has a decisive role.

2. OBJECTIVES OF THE WORK

The original aim of the project (in 2009) was to build a network for material issues related to Gen IV nuclear reactor systems. Since then, the network has evolved to include also other aspects related to Gen IV issues (fuels, reactor physics, thermal hydraulics, waste, safety, etc.). The idea of the network is to combine the resources of different research teams in order to carry out more ambitious and extensive research programs than would be possible for the individual teams. The network is focusing on the following aspects:

- Building of a sustainable Nordic network on GenIV issues, especially focussing on nuclear materials integrity and safety: fuels, cladding, structural materials and coolant interaction.
- Spreading knowledge on GenIV issues and on-going research to different parties involved in the nuclear field, such as LWR licensees universities.
- Inspiring young people to GenIV related R&D.
- Building a strong Nordic forum for GenIV collaboration.
- The results will be used to enhance new project ideas for international programs, e.g., Euratom FP7, calls with a strong Nordic partnership.

3. DELIVERABLES FOR 2012

A seminar (Nordic-Gen4) has been organized 29-31 October in DTU, Risø, in Denmark.

A new website has been made (www.nordic-gen4.org) .

In March 2012, IFE became associate partner to VTT within the Joint Program of Nuclear Materials (see [1]), thereby creating an extra link between the Nordic-Gen4 network and the European programs on nuclear materials.

At IFE/Halden, the development and testing of high temperature LVDT's (up to 700 °C) attained a more mature state and such high temperature LVDTs were delivered the Joint Research Centre (JRC) , showing a good performance (see [2]). The availability of such high temperature LVDTs is very important as these instruments allow online and in-pile measurements on materials and fuels under relevant Generation IV conditions.

Also the development of Electrochemical Potential sensors (ECP) reached maturity and such sensors were sold to JRC, as well as to JAEA in Japan. A collaboration with the Research Centre Rez has also been initiated for qualifying these ECPs under supercritical water conditions (in-pile and out of pile). These sensors can also, with some modification (see [3]) be used as oxygen sensors in liquid metals (sodium, lead, or lead bismuth eutectic (LBE). Collaboration with KTH in Sweden has been initiated for qualifying these sensors in LBE.

The development and testing of coatings for Generation IV reactors is in line with the objectives of the Nordic-Gen4 network, as well as with the European Joint Programme on Nuclear Materials [1]. Coatings under development at Halden [4] have been tested at VTT in supercritical water and at KTH in liquid lead. Such investigations are thus only possible thanks to the availability of various expensive installations within the different Nordic-Gen4 member institutes. A common paper by IFE and VTT on coatings has been submitted to the 6th International Symposium on Supercritical Water-Cooled Reactors (ISSCWR-6), to be held in Shenzhen, Guangdong, China from 3 to 7 March 2013.

Finally, an IAEA Technical Meeting on In-Pile Testing and Instrumentation for Development of Generation-IV Fuels and Materials was hosted by Halden (21 to 24 August 2012). The available contact network of Nordic-Gen4 was very useful in attracting participants to this meeting.

4. ORGANIZATIONS INVOLVED IN NORDIC-GEN4

The partners are shown in Table 2.

Table 2: Partners of NORDIC-GEN4 in 2012

1	Institutt for Energiteknikk (IFE/Halden Reactor Project)	www.ife.no
2	VTT	www.vtt.fi
3	DTU (DTU Nutech)	www.dtu.dk
4	JRC: Joint Research Centre (Petten, The Netherlands)	http://iet.jrc.ec.europa.eu/
5	NRG: Nuclear Services for Energy, Environment & Health (Petten, The Netherlands)	www.nrg.eu/
6	Research Centre Rez	www.cvrez.cz/web/en

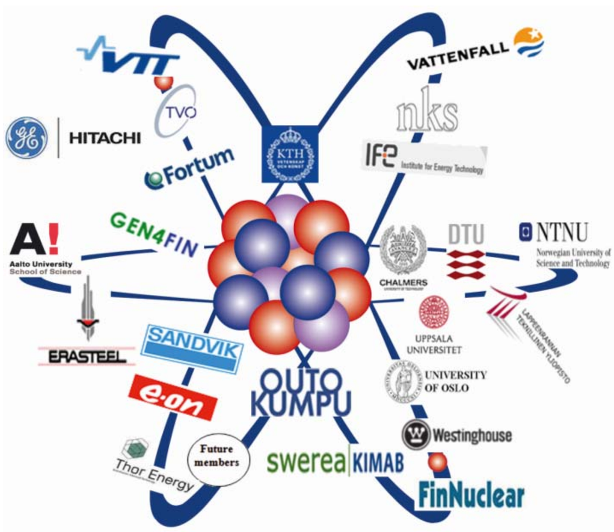

Figure 2: Present Logo for the network

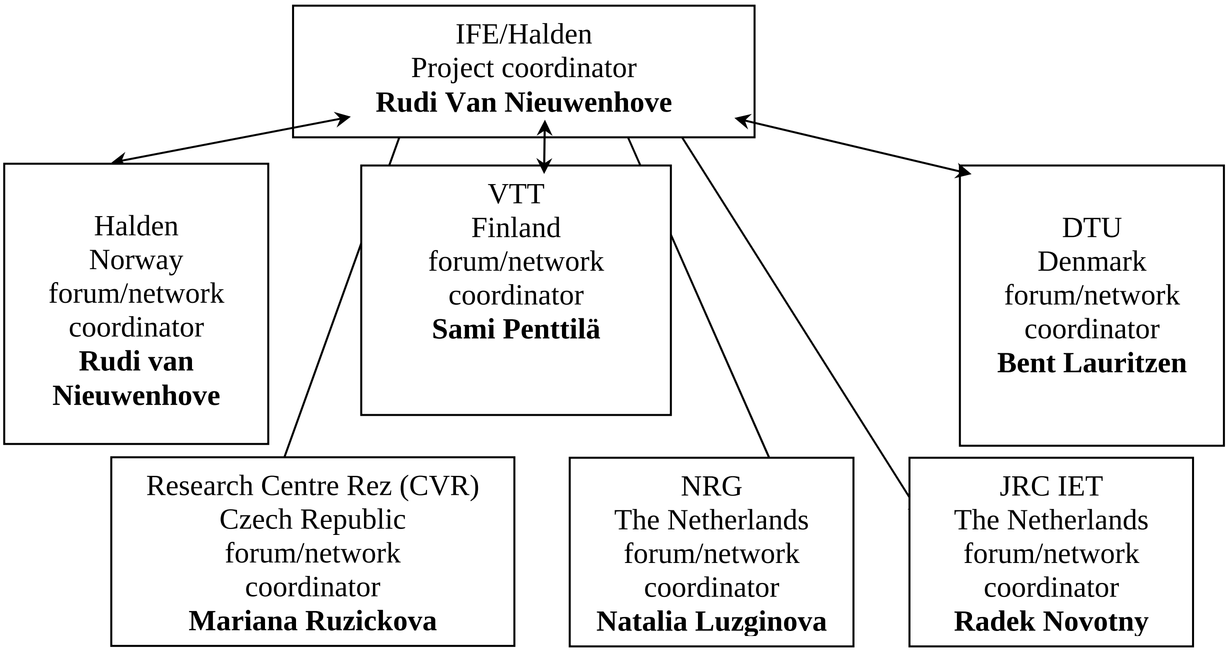

Figure 3: Schematic presentation of the activity organisation for the project NORDIC-GEN4.

The activities of the different partners into Gen-IV has previously been explained in NKS report NKS-254.

The following organizations are members of NORDIC-GEN4 (see www.nordic-gen4.org):

- NKS – Nordic Nuclear Safety Research
- Institute for Energy Technology in Norway, IFE
- GEN4FIN
- JRC IET
- NRG
- Research Centre Rez (CVR)
- DTU Nutech – Center for Nuclear Technologies
- Swedish Centre for Nuclear Technology, SKC
- GE-Hitachi Nuclear Energy
- Studsvik Nuclear AB
- Vattenfall
- EON
- Westinghouse
- Sandvik
- Royal Institute of Technology, KTH
- Chalmers School of Technology
- Uppsala University
- GE-Hitachi Nuclear Energy
- Outokumpu
- Thor Energy, Norway
- University of Oslo
- Norwegian University of Science and Technology

The members of GEN4FIN (see http://virtual.vtt.fi/virtual/gen4fin/members.htm) are given below:

- VTT*
- Fortum*
- Fennovoima*
- TVO*
- Aalto University
- Lappeenranta University of Technology
- Ministry of Employment and the Economy
- Prizztech Oy
- STUK – Radiation and Nuclear Safety
- Tekes – The Finnish Funding Agency for Technology and Innovation

*: Financing through GEN4FIN

5. NORDIC-GEN4 SEMINAR IN DTU NUTECH

A Nordic-Gen4 seminar (2.5 days) was organized by the Center for Nuclear Technologies of the Technical University of Denmark (DTU Nutech). The seminar attracted 49 participants from 12 countries. In addition to the 4 Nordic countries, there were participants from Belgium, The Netherlands, France, Germany, Czech Republic, Poland, Argentina and Ukraine. The participants had the occasion to visit the decommissioning site for Risø's nuclear facilities (Danish Decommissioning). The programme can be found in Appendix A. Based on the reactions during and after the seminar, it can be concluded that the participants were very satisfied. The seminar was funded by NKS.

During the opening session, Bent Lauritzen gave a brief overview of Risø's history and the activities at DTU Nutech. Risø National Laboratory was established in 1958 with the purpose of introducing nuclear energy but in 1985, the Danish government decided that nuclear power should no longer be part of Danish energy planning. At present Denmark is self-sufficient with energy, predominantly due to its oil and gas production which, however, is decreasing, and to its large share of renewables in the primary energy supply. DTU Nutech today carries out research on medical, industrial and environmental applications of nuclear technologies. Activities at DTU Nutech include radioecology and neutronics related to nuclear energy technology and DTU Nutech provides public sector consultancy to the Danish authorities on nuclear emergency preparedness. On this background, the first Nordic-Gen4 seminar in Denmark was welcomed.

Rudi Van Nieuwenhove (present coordinator of the Nordic-Gen4 network) gave a presentation to explain the objectives, structure, functioning and funding of the network.

6. ACKNOWLEDGEMENTS

All the participants to the Nordic-Gen4 seminar are gratefully acknowledged for their contributions. The preparation of the seminar by the organizing team of DTU is greatly appreciated, with special thanks to Dorte Juul Jensen, Søren Fæster, Morten Eldrup, Peter Willendrup and Pia Elhauge.

NKS is acknowledged for funding the Nordic-Gen4 project throughout 2012.

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APPENDIX A: AGENDA OF THE NORDIC-GEN4 SEMINAR IN DTU, RISØ (2012)

Monday, October 29

09:00 – 10:00	**Registration**
10:00 – 10:10	Seminar opening, *Bent Lauritzen*
10:10 – 10:20	Introduction to the Nordic-Gen4 network, *Rudi Van Nieuwenhove*
	Session 1: Generation IV reactor concepts
	Chair: *Concetta Fazio*
10:20 – 10:50	Key Research Challenges for Generation-IV Nuclear Energy Systems *Frank Carre* (invited)
10:50 – 11:20	ELECTRA: European Lead Cooled Training Reactor *Janne Wallenius* (invited)
11:20 – 11:45	*New method of electricity production with using IV generation nuclear reactors* *Marcin Brozek (not presented)*
11:45 – 12:10	A fuel qualification program for ELECTRA *Kyle Johnson*
12:10 – 12:35	Non-proliferation aspects of gen-IV systems. Work being done at Uppsala University *Carl Hellesen*
12:35 – 13:30	**Lunch**
	Chair: *Frank Carre*
13:30 – 13:55	The Myrrha project: current status of design and design challenges *Paul Leysen*
	Session 2: Reactor materials
13:55 – 14:25	Nuclear Materials Research in Europe, present status and future perspectives *Concetta Fazio* (invited)
14:25 – 14:55	Advanced fusion materials characterization: The FEMaS project *Christian Linsmeier* (invited)
14:55 – 15:20	**Coffee break**
	Chair: *Dorte Juul Jensen*
15:20 – 15:45	*Interoperability of Materials Database Systems in Support of the GenIV International Forum* *Tim Austin (not presented)*
15:45 – 16:10	Materials Characterization for Generation IV Reactors at NRG/HFR *Natalia Luzginova*
16:10 – 16:35	Progress in the qualification of candidate materials for Generation IV nuclear systems at the European Commission Joint Research Centre *Krystof Turba*
16:45	*Bus departure for the Viking Ship Museum*
17:00 – 18:00	*Exhibition at the Viking Ship Museum*
19:00	*Reception at Hotel Prindsen*

Tuesday, October 30

	Session 2, cont.
	Chair: *Christian Linsmeier*
08:45 – 09:10	Investigation of coatings for Generation IV reactors *Rudi Van Nieuwenhove*
09:10 – 09:35	Material research for the Generation IV reactors *Eliska Krecanova*
	Session 3: Reactor physics, design and safety
09:35 – 10:05	Status of core components development for Generation IV nuclear energy systems in Ukraine *Mykola Odeychuk* (invited)
10:05 – 10:35	Nuclear reactor system modelling: past, present, and future *Christophe Demazière* (invited)
10:35 – 11:00	**Coffee break**
	Chair: *Ferenc Mezei*
11:00 – 11:25	Development of TALL-3D facility design for coupled CFD and STH code validation *Kaspar Kööp*
11:25 – 11:50	Studies on open-cycle thorium fuel for present light water reactors *Rainer Salomaa*
11:50 – 12:15	Core surveillance and diagnostics of future reactors *Christophe Demazière*
12:15 – 13:15	**Lunch**
	Chair: *Christophe Demazière*
13:15 – 13:40	Neutron noise calculation in a large SFR core and analysis its properties *Zylbersztejn Florian*
13:40 – 14:05	Serpent Monte Carlo code as a modeling tool for GenIV reactors *Jaakko Leppänen*
14:05 – 14:30	Analysis of Uncertainty Propagation in Nuclear Fuel Cycle Scenarios *Karen Atabekjan*
14:30 – 15:00	**Coffee break**
	Session 4: Experimental facilities
	Chair: *Bent Lauritzen*
15:00 – 15:30	The European Spallation Source ESS *Ferenc Mezei* (invited)
15:30 – 15:55	Neutronic design of the ESS target-moderator-reflector system *Luca Zanini*
16:15 – 18:15	**Visit to Danish Decommissioning**
18:30	*Bus departure for Svogerslev Kro*
19:30	*Conference dinner at Svogerslev Kro*

Wednesday, October 31

	Session 4, cont.
	Chair: *Sven Nielsen*
08:45 – 09:10	Danish in-kind simulation efforts toward the ESS, an overview *Esben Klinkby*
09:10 – 09:35	Analysis of steam generator tube leakage in an LFR design *Marti Jeltsov*
09:35 – 10:00	Development and testing of instruments for Generation IV fuel and materials research at the Halden Reactor Project *Rudi Van Nieuwenhove*
10:00 – 10:25	**Coffee break**
	Session 5: Radioactive waste and environmental aspects
	Chair: *Rudi Van Nieuwenhove*
10:25 – 10:50	Fission gas retention in Am containing fuels studied in HELIOS irradiation experiment *Alexander Fedorov*
10:50 – 11:15	Assessment of radiological impact from nuclear installations *Sven Nielsen*
11:15 – 11:40	Transmutation of TRU and long-lived FP elements in an accelerator driven system *Lauri Rintala*
11:40 – 12:05	Production of valuable elements from spent nuclear fuel *Bertel Lohmann Andersen*
12:05 – 12:30	Methods of immobilization of radioactive elements in Perovskite *Tomasz Smolinski*
12:30 – 13:30	**Lunch / end of seminar**

APPENDIX B. BOOK OF ABSTRACTS

Nordic-Gen4 Seminar at Risø, Denmark

29-31 October 2012

4th Nordic Seminar on Generation IV Nuclear Reactors

Book of abstracts

Key Research Challenges for Generation-IV Nuclear Energy Systems

Frank Carre
Scientific Director of Nuclear Energy Division French Alternative Energies and Atomic Energy Commission (CEA)

Fast neutron reactors with a closed fuel cycle have for long materialized the vision of sustainable nuclear energy systems because of their potential for efficiently utilizing Uranium (>80%) and minimizing long-lived radioactive waste. Similarly, high temperature reactors have been early recognized capable of offering extended applications associated with high temperature industrial heat supply. Such advanced nuclear systems that were given a renewed impetus by the Generation IV International Forum, call for scientific and technological breakthroughs in such fields as materials science, heat exchange and fluid dynamics, physical chemistry of reactor systems, nuclear fuel cycle and backend, power conversion systems, as well as instrumentation and controls.

As for LWRs, materials science enables more predictive research in advanced structural materials and fuels that need to resist fast neutron damages and high temperature. Current research in France on sodium cooled fast neutron reactors definitely aims at significant progress in safety and economic competitiveness in order to progress towards a new generation of commercially viable fast reactor of this type. Varied breakthroughs in core design and fuel technology are currently investigated to exclude by design most accidents likely to initiate severe accidents and hence practically exclude accidental large energy release. Progress in safety also drives research in gas power conversion systems that avoid the use of water and subsequent risks of sodium-water interactions. Progress in safety and operability also call for innovations in reactor instrumentation and controls, as well as in techniques for in service inspection and repair. Closing the nuclear fuel cycle of fast neutron reactors enables recycling all nuclear materials generated by LWRs including Plutonium, reprocessed Uranium and depleted Uranium. Minor actinide recycling is an active field of research aiming at assessing the technical feasibility and the additional cost associated with minimizing the long term decay heat and radiotoxic inventory of high level radioactive waste, with potential benefits on non-proliferation.

Along with advancing sodium cooled fast reactor design and technologies, R&D on alternative fast reactor concepts such as gas or lead cooled systems is strategic to overcome technical difficulties and/or political opposition specific to sodium. Such longer term fast reactor designs definitely call for breakthroughs in structural materials, nuclear fuels, reactor systems and power conversion.

High temperature reactors (HTRs) seem to currently yield priority to fast neutron reactors projects that are envisioned as avenues towards sustainable nuclear power. However, current trends of rising fossil fuel prices and prospects of carbon taxes should better value HTR cogeneration products in the medium term and possibly foster the deployment of this type of reactor. Breakthroughs required for next generation HTRs include a small unit power (< 300 MWe) to achieve passive safety features, steel in place of pre-stressed concrete pressure boundary, and gas turbines (in place of steam turbines) for advanced power conversion systems (possibly in direct cycle). Such novel features definitely call for technology advances in structural materials, nuclear fuels, high temperature helium systems and non conventional energy conversion systems (gas turbine, advanced electrolysis, adapted industrial processes and associated heat exchangers…).

Future nuclear energy systems definitely call for enhanced capabilities in multi-scale and multi-physics modelling and numerical simulation. At stake are improving the accuracy of design studies and operating transient analyses, and providing more scientific bases for safety demonstrations that cannot be tested at full-scale such as severe accidents or long term radioactive waste management.

Research and development for nuclear energies (fission and fusion) also benefit from hybridizing to other industrial sectors, as crosscutting science is becoming more important for innovation and direct simulation of basic phenomena. At the same time, increasing opportunities of cross-fertilization exist with advanced technologies developed for non nuclear applications or for fusion devices such as advanced materials, instrumentation, chemical processes and energy conversion systems.

Finally, innovations also call for pre-normative research to support codification and licensing work that will ultimately apply to safety and security features of resulting innovative reactor types and fuel cycles. At all steps of research, international cooperation is a key factor for sharing costs of R&D and supporting facilities, including experimental and prototype reactors.

ELECTRA: European Lead Cooled Training Reactor

Janne Wallenius
KTH Royal Institute of Technology Sweden

Lead cooled fast reactors offer enhanced safety in term of a high boiling temperature, excellent potential for natural convection, good retention of fission products, adequate gamma radiation protection and absence of rapid exothermic reaction with water. The major technological problem is the lack of materials suitable for pump impellers.

An early option to test operational performance of LFRs is application of heat removal by natural convection, thus avoiding the use of pumps in the primary circuit. ELECTRA has been designed by KTH with (Pu,Zr)N fuel, allowing a very small core size with low pressure drop.

KTH, Chalmers and Uppsala university have requested funding from the Swedish government to build the ELECTRA Fuel Cycle Center (ELECTRA-FCC), consisting of a reprocessing facility with a capacity of 4 ton/year, a fuel fabrication facility with a capacity of two pins (360 grams Pu) per day and an 0.5 MW LFR .

In this talk, a general outline of the ELECTRA project is presented, including a plan for R&D necessary to carry out for qualification of fuels, materials and system concept.

New method of electricity production with using IV generation nuclear reactors

Marcin Brozek
Department of Nuclear Physical Chemistry, Institute of Nuclear Physics PAN, Poland

They are forecasting that Liquid Metal Cooled Reactors can works with much higher temperatures, than presently used thermal power plants. Designed temperatures are about 800 – 850 °C (1073 – 1123 K). However, those are, so far, concepts, which needed elaborate adequate materials which can work in such extremal conditions, for many years. Advanced properties of materials are needed not only in steam turbines, but also elements of gas turbines required materials which can stand/resist influent of very hot gases. This same problem should be solved for pipes and conduits which gases are leaded.

Presented are solutions which can improve the efficiency of conversion thermal energy into electricity, and can eliminate parts of problems with destructive influence of moving, hot gases as a medium in nuclear power plants. This innovations concern of new configurations of devices in power stations with IV generations of nuclear reactors.

A fuel qualification program for ELECTRA

Kyle Johnson
KTH Royal Institute of Technology Sweden

In parallel with the research into reactor design and safety currently being performed at KTH for the ELECTRA concept, a fuel qualification program has also been envisaged which encompasses selection of the reference fuel, quality assurance of fuel manufacturing, determination of fuel properties, modelling of fuel behavior, and test irradiations at nominal and off-nominal conditions. While various phases of this program are either currently in progress or have been completed, there remains much work to be done in order to demonstrate fuel safety and reliability, and, ultimately, licensing the fuel for eventual operation.

Non-proliferation aspects of gen-IV systems. Work being done at Uppsala University

Carl Hellesen
Uppsala University, Sweden

Today nuclear power generates about 13-14% of the world electricity production using about 400 reactors. To significantly increase this number and provide the world with abundant clean energy would, in the long haul, likely require the introduction of generation 4 systems and advanced fuel cycles. However, from a nuclear non-proliferation perspective this system would look entirely different than today's. A future with several thousand reactors have been envisaged by some, perhaps with a significant number of smaller modular reactors and so called nuclear batteries with long life cores.

This presentation will review the 4th generation nuclear power from a non-proliferation and safe guards perspective as well as give an overview of some research activities that are done in this field at Uppsala university.

The Myrrha project: current status of design and design challenges

P. Baeten, R. Fernandez, D. De Bruyn, G. Van den Eynde, E. Malambu, P. LEYSEN, H. Aït Abderrahim
Belgian Nuclear Research Centre (SCK•CEN), Belgium

Since 1998, the Belgium Nuclear Research Centre is developing a multipurpose irradiation facility in order to support research programs on fission and fusion reactor structural materials and nuclear fuel development. MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is a flexible experimental accelerator driven system (ADS) with a thermal power of $100MW_{th}$, able to work in both subcritical and critical mode. The Belgian federal government has approved early 2010 the funding of this international project, which from 2023 onwards, will contribute to the development of innovative solutions in the field of nuclear technologies.
In the context of the seventh Framework Programme (FP7), SCK•CEN established a 'Central Design Team' (CDT). Together with 18 European partners from industry, research centres and academia, SCK•CEN worked within the CDT project on the design of a FAst Spectrum Transmutation Experimental Facility (FASTEF). This project has been started on April 01^{st}, 2009 and finished recently end March 2012. The design of the MYRRHA reactor at the end of the CDT project is discussed within this paper. In addition some recent challenges in the design will be discussed. The objectives of this new research reactor are the demonstration of the ADS concept at a reasonable power level on one hand and to proof the technical feasibility of transmutation of minor actinides and long-lived fission products arising from the reprocessing of the radioactive waste on the other hand. MYRRHA will also contribute to the development of the Pb-alloys technology needed for the lead fast reactor (LFR) Technology.
In addition to the demonstration of the efficient and safe transmutation of minor actinides in ADS systems, the implementation of the MYRRHA project will allow for:

- Fuel developments for innovative reactor systems;
- Material developments for GEN IV systems and fusion reactors;
- Doped silicon production;
- Radio-isotope production;
- Fundamental science application at the high power proton accelerator.

In this paper we will report on the status of the project.

Nuclear Materials Research in Europe, present status and future perspectives

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Since 2009 several European Research Associations have launched the Joint Program Nuclear Materials (JPNM) in the frame of the European Energy Research Alliance. The JPNM addresses nuclear materials issues related to the advanced nuclear systems as developed in the frame of the SET-Plan industrial initiative for nuclear energy. These advanced nuclear systems address the sustainability of nuclear energy, where resources preservation and waste minimization are prominent topics addressed. Moreover, safety aspects have top priority for the development of these systems. In this context, the development and qualification of innovative materials that can withstand a more demanding environment and comply with safety requirements are considered mandatory. The JPNM addresses these aspects through four dedicated sub-programs. The sub-programs have been structured such as to qualify conventional steels and to develop and qualify innovative materials as e.g. SiCSiC and ODS. Further, a multiscale modeling sub-program with the objective to enhance predictive capability of materials behavior in harsh environments is as well included. The activities of the JPNM are strongly linked to several national programs and EC funded projects as e.g. GETMAT, MATTER, MatISSE.
The present contribution will give an overview on the current status of the JPNM and highlight key motivations that have brought to the present JPNM structure. Moreover, selected and relevant results from JPNM and the EC funded project will be integrated in this contribution with the objective to discuss key materials items and future perspectives to advance in this relevant area.

Advanced fusion materials characterization: The FEMaS project

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For the realization of fusion as an energy source, the development of suitable materials is one of the most critical issues. The required material properties are in many aspects unique compared to the existing solutions, yet alone due to the necessary resistance to irradiation with neutrons at energies up to 14 MeV. In addition to deterioration effects due to neutrons, the mechanical stability of structural materials has to be maintained up to high temperatures. Plasma-exposed materials must be compatible with the fusion plasma, both for impurity generation to the plasma and resistance to erosion and hydrogen isotope retention. The development of materials fulfilling these and other criteria is a long-term activity which involves basic materials science, materials development, characterization both under loading conditions and off-line, as well as testing under neutron loading conditions.
For the realization of a DEMO power plant, the materials solutions must be available in time. The European initiative FEMaS – Fusion Energy Materials Science – aims at accelerating materials development by integrating advanced materials characterization techniques, among them the efficient use of neutron and synchrotron-based techniques, into the fusion materials community. Further, high-end transmission electron microscopy and mechanical characterization (including on a microscopic level in order to facilitate tests of small material

volumes, e.g. from neutron irradiation campaigns) are to be more deeply applied in fusion materials research. Finally, irradiation facilities for neutron damage benchmarking are contributing to the understanding of radiation effects.
The presentation will demonstrate the recent advancements in fusion materials research, e.g. by applying synchrotron and neutron tomography to novel materials and components. Deeper understanding of radiation effects is achieved by in situ TEM of materials under irradiation. Modeling of irradiation effects is linked to activities at irradiation facilities. Finally, new developments in mechanical testing at micro- and nano-scales are presented.

Interoperability of Materials Database Systems in Support of the GenIV International Forum

Tim Austin
European Commission JRC Institute for Energy and Transport

Database applications for engineering materials contribute to the research process and support the more responsible and effective use of resources invested in the development of materials for nuclear applications.

Across all disciplines, there is an ever increasing demand to improve transparency and make data available—publishing houses are placing more stringent demands on authors to make their data accessible and funding agencies are stipulating that data plans accompany project proposals. This includes the European Commission, which with its Communication on open data as an engine for innovation, growth and transparent governance, is encouraging the conservation and exchange of public sector information (PSI), including the results of publically funded research. Together with these external factors, increasing competition for funds, there is an ever more frequent expectation that research consortia reuse materials data generated during former research programmes.

Motivated by these factors and in an effort to support the data and knowledge management needs of the GenIV International Forum (GIF), the European Commission Institute for Energy and Transport (IET) and Oak Ridge National Laboratory (ORNL) are collaborating on a systems interoperability project that involves the materials database applications hosted at each institute—MatDB Online and the GenIV Handbook, respectively. MatDB Online is a database for engineering materials test data that has benefited from many hundreds of person years invested in the development of a robust data model, comprehensive test support, and an intuitive user interface. It contains over 20.000 test results and provides a web-interface for data content, data entry, data retrieval, and analysis routines. The database covers mechanical and thermo-physical properties data of engineering alloys generated in accordance with international material standards and recommendations. The GenIV Handbook is built on a commercial application that relies on a hierarchical/relational model for navigating data. For example, starting from a materials record, a tree like navigation frame allows a specific material specification to be located, which itself incorporates links to pedigree records and test data. These links are maintained by the system based on criteria (i.e. relations) between records. MatDB and the GenIV Handbook have been developed wholly independently, and so while both applications store materials data, their data models are very different.

The work to enable exchange of data between the systems leverages the fact that both applications have interfaces for XML data import and export. The use of XML as a messaging format is not sufficient however to achieve systems interoperability. Additionally, Standard representations of materials test data are needed that allow local data models to be transformed to a recognised intermediate format. To this end, IET and ORNL are leveraging Standards for engineering materials test data that are emerging from a series of CEN Workshops, the latest of which is CEN/WS SERES (Standards for Electronic Reporting in the Engineering Sector).

This paper elaborates on the purpose, features, and contents of the MatDB and GenIV Handbook applications and describes the process whereby emerging Standards for engineering materials test data are being leveraged to address systems interoperability.

Materials Characterization for Generation IV Reactors at NRG/HFR project

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The Nuclear Research and consultancy group (NRG) is the center of nuclear expertise in the Netherlands. NRG develops knowledge, processes and products for safe application of nuclear technology in the areas of energy, environment, and health. This includes technology development for safe and economic operation of future nuclear energy systems, such as Generation IV reactors. NRG performs a range of fuel and material irradiations at the High Flux Reactor in Petten (the Netherlands).

The High Flux Reactor (HFR) is a powerful multi-purpose research and materials testing reactor operating for about 280 Full Power Days per year. In combination with the hot cells facilities, HFR provides irradiation and post-irradiation examination services requested by nuclear energy research and development programs, as well as by industry and research organizations. Using a variety of the custom developed irradiation devices and a large experience in executing irradiation experiments, the HFR is suitable for fuel, materials and components testing for different reactor types. Irradiation experiments carried out at the HFR is mainly focused on the understanding and prediction as well as on monitoring and mitigation of the irradiation effects on materials.

The present paper will describe the main NRG activities and highlight the implications on characterization and qualification of nuclear structural materials for Generation IV reactors.

Progress in the qualification of candidate materials for Generation IV nuclear systems at the European Commission Joint Research Centre (JRC-IET

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European Commission Joint Research Centre, Institute for Energy and Transport

Advanced materials envisaged for use in Generation IV nuclear systems will be subject to more extreme environments than those employed in currently operating reactors, both in terms of temperature and neutron dose but also with regard to coolant compatibility. In the current stage of development, it is of key importance to carry out the relevant material performance assessment in order to demonstrate that the candidate materials meet the design objectives.
The European Commission Joint Research Centre (JRC) Institute for Energy and Transport (IET) is participating in the 7th Framework Programme research project GETMAT, "Generation IV and Transmutation Materials". So far, the institute has concentrated mainly on the creep performance assessment of the GETMAT 14Cr oxide dispersion strengthened (ODS) ferritic steel, using both conventional uniaxial creep tests and small punch (SP) creep testing. Low cycle fatigue tests with holding times are planned to assess the creep-fatigue interaction of the alloy. The small specimen size for the small punch creep testing made it possible to extract specimens from the extruded ODS steel rod with two orientations, and thus to evaluate the anisotropy in the material's creep resistance. Creep tests, both uniaxial and small punch, were carried out at 650 °C with rupture times up to several thousand hours. It was found that the 14Cr ODS alloy exhibits extremely strong anisotropy in creep resistance.
At a given stress, the extrapolated rupture time is of the order of years when the material is loaded parallel to the extrusion axis, but less than one hour when loaded in the transverse direction. The anisotropy was also observed using scanning electron microscopy (SEM) on fractured SP specimens, both in the appearance of macroscopic cracking patterns and in features of the fracture surfaces. Even for the longitudinal orientation (parallel with the extrusion direction), the alloy's creep performance at 650 °C is inferior to the expectations of the manufacturer (CEA). It can be concluded that if the material is to be used in Gen IV systems, the corresponding design codes would need to take into account its transverse properties rather than the (more easily obtainable) longitudinal ones.
The direct contribution of the JRC-IET to the Nordic-Gen4 network for 2012 includes the development of the devices for SCC crack growth rate measurements in LWR and SCWR conditions using a pneumatic bellows based loading device. Increased cooperation in the framework of Nordic-Gen4 is demonstrated by the development of a new high temperature and pressure LVDT sensor and reference electrode operating up to SCW conditions in the IFE OECD Halden Reactor Project, which were successfully incorporated into the SCC testing devices at the JRC-IET. Further cooperation will include the development of sensors for a future IASCC in-pile testing device, as well as sensors operational in the liquid Pb/Bi eutectic. The conceptual design for the future IASCC application and for the pneumatically powered testing system capable of functioning in a liquid lead and Pb/Bi environment up to 650 ºC is already available and will be presented.

Investigation of coatings for Generation IV reactors

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Plasma coatings, applied by physical vapour deposition, as well as plasma surface treatments could find interesting applications in the nuclear field, both for existing type of reactors as well as for generation-IV reactors. Among the possibilities are; hardening of components, protection against wear, reduction of tritium diffusion, reduction of hydrogen embrittlement, reduction of friction and providing electric insulation. A large variety of industrially available coatings exists. Among these, the following coatings (PVD) were investigated: TiAlN, CrN and ZrO_2. Furthermore, pulsed plasma surface treatment (PPST) was investigated as well. Small coated samples were exposed to supercritical water (SCW) at 650 °C and to liquid lead at 550 °C. For supercritical water, the CrN coating was found to provide a stable corrosion barrier. Scanning Electron Microscope (SEM) images of the coatings have been obtained before and after the exposures. In SCW, it was found that a chromium oxide layer forms on the surface of the CrN coating. For liquid lead, ZrO_2 was found to be the most promising coating. PPST was found to harden significantly the surface of zircaloy samples and could therefore reduce fretting corrosion of fuel rods. The effect of various coatings on the hydrogen diffusion through metals has also been investigated. ZrO_2 coatings were found to decrease significantly the diffusion of hydrogen (and tritium) through Inconel 600. In those types of generation-IV reactors where fuel rod claddings cannot be made out of zircaloy (because of too severe corrosion) such coatings might become indispensable for preventing tritium release from fuel rods into the coolant.
In addition to the investigation of coatings, it has also been found that AISI 410 has good corrosion resistance in liquid lead/bismuth, especially when combined with Zr as inhibitor. This can be attributed to the formation of a thin ZrN protective layer. Results of long duration (1000 h) corrosion tests, at high temperature (up to 750 °C) will be presented.

Material research for the Generation IV reactors

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Research centre Rez, Czech Republic

Research centre Rez deals with material research for GenIV reactors cooled with heavy liquid metals, supercritical water and gas (helium) (LFR, SCWR, VHTR/GFR, respectively). It deals also with the GenII, GenIII reactors and fusion. At the moment we take care about classic fossil-fueled blocks of non-nuclear power plants too.
GenII, GenIII research studies irradiation and radiation damage to monitor the development of the damage due to material properties. Examines IASCC resistance and crack propagation speed in order to predict the resistance due to the time of operation; corrosion resistance due to guidelines for nuclear power plants; experimental and calculation setting of the neutron characteristics of absorption materials with boron or other additives for the reason of material properties verification and last but not least the fuel cover and coolant interaction to study oxidation / layers depending on the water chemistry.
LFR research deals with corrosion resistant materials and coatings testing and liquid metal embrittlement to search suitable construction materials. SCWR cooled reactors research takes

care about behavior of water under supercritical parameters, influence of water chemistry, radiolysis of SCW; influence of SCWR environment on mechanical properties and corrosion resistance of constructional materials for SCWR. VHTR/GFR research handles with impurity caused high-temperature corrosion; phase and structural stability and helium purification technology.

Status of core components development for Generation IV Nuclear Energy Systems in Ukraine

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In NSC KIPT are created and proved designs and manufacturing technologies of fuels and core components for the Generation IV nuclear energy systems – High Temperature Gas- cooled Reactors (HTGRs): VGR-50, VGM, VG-400; Gas Reactors on Fast neutrons: BGR and BRGD; high-temperature Molten Salt nuclear Reactor (MSR); other special reactors.
The special attention is given to discussion of the basic technological, reactor test schemes and main properties for fuels and some core components. Results of performance capacity researches of fuel elements and their components, structural materials in reactor irradiation conditions, are discussed: temperature up to 1600 ^{o}C, burn-up up to 20 % fima.

Nuclear reactor system modeling: past, present, and future

Christophe Demazière
Chalmers University of Technology, Sweden

The simulation of the behavior of nuclear reactor systems is a challenging task in complex system modeling, with the complexity residing in the various physical and geometrical scales that such systems present, as well as in the interdependency between different transport phenomena. This is particularly true for the part of the system containing the nuclear core, where neutron transport, fluid transport, and heat transfer are three key phenomena to be simultaneously tackled. In addition, other physical phenomena than the ones mentioned (such as fluid-structure interactions, fuel rod behaviour, etc.) might also be of importance. As such, nuclear systems are often referred to as multi-physics and multi-scale systems. With the rapid development of computer technology in the seventies, different modeling strategies have emerged. In the area of neutron transport, two basic approaches exist: either a stochastic approach (i.e. Monte Carlo) where the behavior of neutrons is described from the probabilities of occurrence of any nuclear reaction type on any species at any energy, or a deterministic approach where balance equations expressing the conservation of the number of neutrons in given volumes, solid angles, energy bins at any time are solved. In fluid transport and heat transfer, an area commonly referred to as thermal-hydraulics in nuclear engineering, equations expressing the conservation of mass, linear momentum, and energy are averaged on rather coarse volumes and time bins, and thereafter solved (system code approach). With the latest developments in computing capabilities, these macroscopic approaches are being complemented by “mesoscopic” approaches (usually referred to as Computational Fluid Dynamics or CFD)

where much smaller volumes and time bins are considered. Each of the aforementioned tools and approaches has its own area of validity, and advantages/disadvantages.
From a computational viewpoint, deterministic neutron transport and system code approaches are fast running methods specifically targeted at allowing the simulation of non-steady-state conditions and at performing safety evaluations. The resolution of the different physical scales on the neutron transport side is achieved by the successive computation at different levels, starting from the meso-scales (pin cell level) up to the macro-scales (full core calculations). On the thermal-hydraulic side, the possibility of modeling large systems arises from the fact that both small scales and high frequency phenomena are filtered out, and reintroduced artificially by using empirically-derived closure relationships. The presentation will explain the main underlying principles presently used in the modeling and simulation of nuclear reactors, and will also give an overview of the current efforts aimed at circumventing the limitations of such methodologies.

Development of TALL-3D facility design for coupled CFD and STH code validation

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Royal Institute of Technology, Sweden

Next generation pool type nuclear reactors are aiming towards simplicity and economic competitiveness. It can be shown that 1D system codes, used for reactor safety analysis, are generally not able to resolve complex transients with mixing and stratification while 3D codes are computationally too expensive for solving the whole primary coolant system. The goal of code coupling is to achieve necessary accuracy with affordable computational efficiency. TALL-3D facility is designed to have integral feedbacks between the 3D test section and the system and to produce data for coupled code validation. The design process of TALL-3D facility and proposed coupling methodology is presented.

Studies on open-cycle thorium fuel for present light water reactors

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Thorium is an interesting alternative to supplement uranium resources. Furthermore, the Th-U cycle may offer remarkable advantages as compared to U-Pu cycles. The very long transition to a pure Th232-U233 system, however, requires several intermediate reactor types involving all the nuclides U235, U238 and Pu239. Th-fuel cycle can be implemented only in large nuclear countries or within politically accepted collaborations. Here we consider a highly simplified case where Th-fuel bundles are introduced into present-type LWR cores and analyze how these would perform. These studies possess education and training advantages, enable validation of reactor codes in non-standard parameter ranges, and provide critical assessment on benefits of thorium fuel.
We discuss nuclide generation in thorium containing nuclear fuel from the viewpoint of safeguarding and long-lived nuclear waste. Several burn-up code packages (CASMO, Serpent, Monteburns, and DeTra) to calculate nuclide inventories have been used and further developed

to improve their accuracy and speed. The goal is to make quantitative estimates for the waste toxicity of spent fuel options.
The applicability of thorium based fuel in present LWRs has been explored considering both single assemblies and/or full core loadings. The Radkowsky seed-blanket configuration in PWR has been assumed. In a BWR, more fuel alternatives enter because of the 3D variation of core composition and neutron spectrum. In both PWR and BWR cores, individual thorium containing fuel bundles appear technically feasible. Neutronics calculations have involved various Monte Carlo methods (MCNP, Serpent) and up to date neutron transport code packages (CASMO-4E and SIMULATE-3). We have also used these tools for modeling advanced lead cooled fast reactor configurations. Large burn-up, fast neutron spectrum and possible new coolants make the materials performance of the thorium based fuel particularly demanding.

Core surveillance and diagnostics of future reactors

Christophe Demazière
Chalmers University of Technology, Sweden

The Division of Nuclear Engineering at Chalmers University of Technology is one of the pioneering institutions in the area of core surveillance and diagnostics. Assuming steady-state conditions, the measurement of process signals using the existing instrumentation supplemented by adequate data acquisition chains allows monitoring stationary fluctuations. Even though the system is running at steady-state conditions, these fluctuations carry some information about the dynamics of the system, and can be used either for core diagnostics/surveillance purposes (i.e. when an anomaly is suspected in the core) or for determining dynamical core parameters/safety coefficients. The main advantage of such techniques relies on the fact that no perturbation of the system is required and that the method is thus a non-intrusive one.
Numerous noise diagnostics tasks have been successfully developed and tested in LWRs. In most cases, these tasks imply inferring from the measurement of a few process parameters the origin of the observed fluctuations. Such a process is usually referred to as “unfolding”. This unfolding requires the prior determination of the reactor transfer function, i.e. the function allowing estimating the neutron fluctuations (also called neutron noise) induced by given known fluctuations (usually referred to as noise source).
In a current effort to extent noise diagnostics to fast reactor systems, the Division of Nuclear Engineering at Chalmers University of Technology has been pursuing two main lines.
The first one corresponds to the understanding of the dynamical behavior of fast systems which significantly differs from thermal ones. For that purpose, and in order to get physical insight, simple enough configurations have to be considered so that analytical solutions can be determined. Some examples concerning MSRs will be presented, where the circulation of the fuel is responsible for some interesting and new dynamical features.
The second line of research lies with the work initiated at the Division to estimate the open-loop/closed-loop reactor transfer functions for three-dimensional fully heterogeneous systems. A numerical tool has for instance been developed for estimating the closed-loop transfer function of PWRs. Some examples will be presented about the extension of the tool to fast reactor systems, and notably the ESFR. Although the tool is currently limited to the determination of the open-loop transfer function, this tool is the only one of its kind, and brings some new insight about multi-group reactor dynamics of fast systems.

Neutron noise calculation in a large SFR core and analysis its properties

Zylbersztejn Florian
Chalmers University of Technology, Sweden

The purpose of the presentation is to expose the results of the recent neutron noise deterministic calculation realized on the ESFR (European sodium fast reactor) core model. The CORESIM calculation code, based on diffusion theory with multi-energy groups, has been created to solve the neutron noise equations on a hexagonal geometry meshing.
A formalism of the equation describing the nuclear reactor dynamics adapted to the neutron noise calculation will be presented, showing the possibilities of calculation of the neutron noise on a system at steady state by a model of the initiating phenomena expressed as a perturbation of the macroscopic cross sections.
The 3 dimensional and 33 energy groups ESFR model used for the CORESIM calculations, has also been evaluated with the ERANOS code, which is a reference for fast system calculations. The ERANOS transport code has provided a comparison tool for the static flux calculation, allowing a verification of the solving algorithms, and an evaluation of the bias between the diffusion and transport calculations.
The results of the calculations are the amplitude of the neutron noise and the phase with regard to the source. Using first order perturbation theory, it has been possible to obtain very interesting characteristics of the neutron noise, in term of the spatial shape as a function of the frequency of the source, and the amplitude and the spatial relaxation length of the local components.

Serpent Monte Carlo code as a modeling tool for Gen IV reactors

Jaakko Leppänen
VTT Technical Research Centre of Finland

The Serpent Monte Carlo reactor physics burnup calculation code has been developed at VTT Technical Research Centre of Finland since 2004. The code is distributed by two data centers - OECD/NEA Data Bank and RSICC, and is currently used in 67 universities in 27 countries around the world. Typical user applications range from group constant generation to fuel cycle studies and research reactor modeling.

This presentations gives an overview on the advantages and drawbacks of using the continuous energy Monte Carlo method for the modeling of generation-IV reactor systems, with a particular emphasis on liquid metal-cooled fast reactors and high-temperature reactors with randomly distributed particle fuel.

Analysis of Uncertainty Propagation in Nuclear Fuel Cycle Scenarios

Karen Atabekjan
Royal Institute of Technology, Sweden

The talk will be focused on my experience contributing a summer project on analysis of uncertainty propagation in nuclear fuel cycle scenarios (NFC). This work was performed as a student of University of Tartu (UT), in the framework of an Erasmus practical internship at the Transmutation team (TRANS) of the Institute for Nuclear and Energy Technologies (IKET), Karlsruhe Institute of Technology (KIT) during the period 24.05.2012 – 12.07.2012. In recent years, NFC analysis has become the field of interest of the nuclear energy group of University of Tartu, focusing on studying environmental and economic impacts of different scenarios on Estonia.

NFC is a complex system describing the stages and facilities that accompany the whole process of producing energy utilizing fissile materials and the fission reaction. NFC covers all the stages, starting from ore mining and ending with waste disposal. One of the areas where simulations of NFC scenarios can provide some useful input is the introduction of advanced reactors (fast critical or accelerator-driven systems) in order to substantially reduce the burden of nuclear waste.

The performance of transmutation technology can be assessed with some integral quantities (such as neutron and gamma emission, decay heat, radiotoxicity), all at different stages of fuel cycle. For this project, a specific scenario was considered, shortly described by the need to get rid of the current stock and stabilize the further production of plutonium and minor actinides produced by currently operating light water reactors.

The main interest of this study was to assess the influence of uncertainties in the values of cross sections on some of the output results of NFC analysis. The work on the topic consisted of several steps. Firstly, the radionuclides of particular interest were determined. Secondly, changing the initial cross section values by a fixed percentage, the sets of reduced and increased values were obtained. For this purpose, a script was written in Octave, which served as an intermediate link between the neutronic code ERANOS and the NFC analysis code COSI. Writing and testing the aforementioned script constituted most of the summer project work. For the simulation, a fast reactor with Conversion Ratio 0.5 and Minor Actinides to plutonium ratio of 1 was considered. After some further processing, mainly consisting in collapsing 33-groups cross sections produced by the neutronic code into 1-group cross sections required by the scenario code, the modified data was fed into COSI. Then, from the changes of output values, the corresponding sensitivity coefficients were calculated.

The European Spallation Source ESS

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European Spallation Source ESS AB, Lund, Sweden

ESS will be a linear accelerator driven long pulse spallation source of 5 MW average, 125 MW peak proton beam power. The neutron generation in the target and its safety have several features relevant for or related to aspects of nuclear power generation. The basic concepts of ESS and its target design will be presented.

Neutron design of the ESS target-moderator-reflector system

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Starting operations in 2019, the European Spallation Source (ESS) will be a pulsed neutron source with an unprecedented brightness, thanks to a 5 MW proton beam (2.5 GeV at 2 mA) impinging on a high-density target.
At present the project is in the design update phase, aimed at reaching by the end of 2012 a technical design, where a sound solution will be discussed in detail. The neutronic design of the target-moderator-reflector (TMR) system is at the core of the work, with several areas of study also of interest to the reactor community, such as: gaining experiences from existing neutron sources (spallation sources and research reactors); investigation of libraries and models, to make the best use of existing knowledge and tools in the Monte Carlo (MCNPX) simulations; use of advanced techniques, and development of new software tools for the modeling of neutron transport, with applications to beam extraction and target-moderator design optimization. In this talk and overview of the main activities is given, and the TMR design is presented.

Danish in-kind simulation efforts toward the ESS, an overview

Esben Klinkby
DTU Nutech, Technical University of Denmark, Risø Campus

A consortium of Danish university groups participate in simulation efforts toward the future European Spallation Source in Lund, Sweden. The presentation will discuss various computational tools and cover how we use these to model all design aspects of the cold and thermal neutron beam lines from moderator surface through neutron optics to the experimental stations.

Analysis of steam generator tube leakage in a LFR design

Marti Jeltsov
Royal Institute of Technology, Sweden

Pool-type design of a lead cooled fast reactor (LFR) has a number of safety related and economic advantages. Still, erosive and corrosive coolant environment poses a threat to integrity of steam generator tubes (SGTs). There is probability that small steam bubbles leaking from a SGT will be dragged by the coolant flow into the core region causing a threat of reactivity insertion accident or local damage of the fuel rods. CFD analysis of the likelihood and consequences of such events is presented.
Development of an experimental facility to validate numerical models for bubble transport in LBE is described.

Development and testing of instruments for Generation IV fuel and materials research at the Halden Reactor Project

Rudi Van Nieuwenhove
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Generation IV (Gen IV) nuclear fission technology aims at the development of more sustainable nuclear power, including better utilization of uranium resources, minimization of radioactive waste, as well as improved economics and safety.
In the European Strategic Energy Technology Plan (SET-Plan), maintaining European competitiveness in fission technologies and the development of Gen IV reactors is a priority for the long-term energy planning. Challenges for materials selection for Gen IV nuclear reactors are related to maximization of the nuclear fuel efficiency and minimization of the disposed wastes. Furthermore, the challenges are identification, development and testing of materials and fuels which can be used at higher temperatures, in more corrosive environments and to higher neutron fluences. This implies the need to develop instruments which can operate under these demanding conditions. One of the basic instruments to study fuels and materials is the Linear Variable Displacement Transducer (LVDT), since this instrument can be used to measure fission gas release (fuel rod pressure), cladding elongation and fuel swelling, fuel centre temperature and creep of materials. An LVDT which can be used up to 550 °C has recently been developed at the Halden Reactor Project and a first prototype has been made which can be used up to 900 °C. For materials studies in supercritical water (SCW), an electrochemical potential sensor has been developed which can be used up to 650 °C and tests have been performed at the SCW loop at VTT (Finland) showing good performance. Preparations are under way to use the same type of sensor also as oxygen sensor in liquid lead.

Fission gas retention in Am containing fuels studied in HELIOS irradiation experiment

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The HELIOS irradiation experiment has been performed at High Flux Reactor in Petten as a part of the EUROTRANS 6th Framework Project. The objective of the HELIOS experiment was to study the in-pile behavior of uranium free inert matrix fuels foreseen for transmutation of minor actinides. The HELIOS irradiation has been carried out in two channels containing in total 5 sealed pins with the following materials: 1)MgO+ Am2Zr2O7 CerCer composite pellets; 2) (Am,Zr,Y)O2 solid solution pellets; 3) (Pu,Am,Zr,Y)O2, solid solution pellets; 4) (Am,Zr,Y)O2+Mo CerMet composite pellets; 5) (Pu,Am)O2+Mo, CerMet composite pellets. The reason to add Pu in pins 3) and 5) was to increase the irradiation temperature through nuclear heating, so that separate effect of the temperature on fission gas retention could be studied.
The fuels were irradiated for 241 full power days, after which the Post Irradiation Examination (PIE) was carried out in the Hot Cell Laboratories of NRG. Within the PIE program the capsules were punctured and the pressure build up in the capsules due to release of the fission gasses was measured. Then the gas recovered from the capsules was analyzed with mass spectrometer to determine the elemental and isotope composition. By comparing the absolute amounts of measured Helium, Krypton and Xenon with the produced amounts obtained from FISPACT calculations, the corresponding released fractions for every element were calculated. The effect of the irradiation temperature on the released fractions was clearly observed. The findings on the fission gas release were correlated with optical and scanning electron microscopy results.

Assessment of radiological impact from nuclear installations

Sven Nielsen
DTU Nutech, Technical University of Denmark, Risø Campus

Radioecological research started in Denmark in the late 1950’s on radioactive fallout from nuclear weapons testing. The results were used to assess radiological consequences of a hypothetical accident at the Barsebäck nuclear power plant located 2 km from Copenhagen. The results were also used to make a quick and robust assessment of the radiological impact in Denmark of fallout from the Chernobyl accident. Compartmental modelling was used to assess the impact of radioactivity in the Baltic Sea. Contemporary models for transfer of radioactivity through human foodchains were developed from experience on fallout from nuclear weapons testing and nuclear reactor accidents. These models are now applied to analyse the radiological impact from routine operation and hypothetical accidents of ESS.

Transmutation of TRU and long-lived FP Elements in an Accelerator Driven System

Lauri Rintala
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The spent nuclear fuel is one of the key aspects for the future of nuclear energy. Spent fuel remains highly radioactive and toxic for over 100 000 years. The extremely long time scale exceeds our detailed technological knowledge when deep geological disposal is considered and induces huge uncertainties that impede licensing of disposal and gaining public acceptance.
After 500 years cooling most of the fission products (FP) have decayed and the main sources of toxicity in spent fuel are uranium and plutonium isotopes since they are also most abundant. Uranium and plutonium can be removed and reused, preferably, in a fast reactor or once in a thermal reactor. Without uranium and plutonium the main contributors for long term toxicity are long-lived FPs I-129 and Tc-99, and minor actinides (MA), mainly americium and curium.
MAs and FPs introduce parasitic absorption and impair temperature feedbacks of nuclear fuel and thus re-inserting them into a critical reactor is undesirable. Our solution to these issues is incineration in specially designed subcritical accelerator driven systems (ADS).
Belgian MYRRHA concept is a most prominent ADS in the world with planned start of full operation in 2022. It is designed to have a 600 MeV and 3.5mA proton beam with lead-bismuth-eutectic (LBE) spallation target and MOX fuel. LBE coolant provides fast spectrum for material research. In this study the MYRRHA core is simulated using Monte Carlo codes FLUKA and Serpent. In addition to the standard MYRRHA design, also depleted uranium is considered for spallation target material and thorium-bearing materials for fuel.
The depleted uranium target outperforms the LBE target in neutron production, but it also increases the minor actinide production and thus the total MA balance becomes less negative. Thorium is used in Th-Pu and Th-233U fuels. The usage of thorium reduces additional MA production but slightly lowers breeding, i.e., the production of fissile material.
The materials to be transmuted were inserted into the core by substituting fuel in a fuel rod with pure 129I or 99Tc, or AmO2. The actual transmutation rate observed in simulations is rather modest, e.g., for technetium and iodine the rate is circa 0.04%/day. Hence, with irradiation time of 1200 days almost half of initial inventory would be transmuted.

Production of valuable elements from spent nuclear fuel

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Nuclear fission fragments with high yields have mass numbers in the intervals 88-103 and 130-145. It appears that 55 fission fragments are stable and 19 are radioactive with $T_{½} < 30$ days. Together these 74 elements will be stable in spent nuclear fuel after a few years. In this paper it is discussed whether it might be attractive in some future to extract valuable elements from spent nuclear fuel. A very preliminary estimate indicates that fission of one ton of U-235 might produce about 140 kg of neodymium. In a distant future when spent nuclear fuel is processed for other reasons it may be profitable to extract this valuable element and maybe other elements.

Methods of immobilization of radioactive elements in Perovskite

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Synroc (Synthetic Rocks) materials have been regarded as the second generation of high level waste (HLW) forms in the world. It allows to incorporate into their crystal structures almost all of the elements present in high-level radioactive waste.
One of the components of Synroc-C is perovskite ($CaTiO_3$) which immobilize mainly fission products, but also allow to immobilize in his structure long-lived actinides such as plutonium (Pu). Perovskite phase has been fabricated by a sol-gel route. In the present work our proprietary complex sol gel process (CSGP Polish Patent PL 172618, 1997) and method of synthesis Me-titanates (Polish Patent PL 198039, 2001) were adapted to prepare of perovskite. Additions of 10% molar Sr, Co, Cs and Nd into Ti-Ca-nitrate sols were carried out by CSGP. Gels obtained by evaporation of sols under reduced pressure were thermal treated according thermogravimetric (TG) analysis. Differential Thermal Analysis. X-ray diffraction and IR studies indicated that transformation of ascorbate- nitrate gels into doped orthorhombic perovskite phases is definitely lower (about 650^oC) than those pure nitrate gels (approx. 700^oC). It means that nuclear wastes surrogates were homogenously distributed into crystalline structure of perovskite and elaborated process can be applied for HLW immobilization.

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Abstract	The Nordic-Gen4 (continuation from NOMAGE4) seminar was this year hosted by DTU Nutech at Risø, Denmark. The seminar was well attended (49 participants from 12 countries). The presentations covered many aspects in Gen-IV reactor research and gave a good overview of the activities within this field at the various institutes and universities. Participants had the possibility to visit the Danish Decommissioning site.

Key words	Nordic-Gen4, seminar, Risø

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